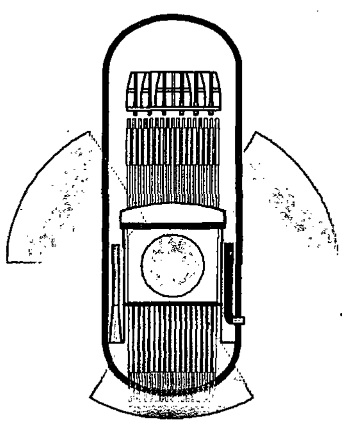
Attachment 4
James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 14

Fluence Calculation
(Non-Proprietary Version)

NON-PROPRIETARY VERSION OF JAMES A. FITZPATRICK REACTOR PRESSURE VESSEL FLUENCE EVALUATION AT END OF CYCLE 17 AND 54 EFPY



PREPARED FOR
ENTERGY NUCLEAR
FITZPATRICK, LLC
OCTOBER 2007

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NON-PROPRIETARY VERSION OF JAMES A. FITZPATRICK REACTOR PRESSURE VESSEL FLUENCE EVALUATION AT END OF CYCLE 17 AND 54 EFPY

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CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

This report presents the results of the reactor pressure vessel (RPV) fluence evaluation performed for the James A. FitzPatrick Nuclear Power Station ('JAF'). Reactor pressure vessel fluence is determined for energy >1.0 MeV and energy >0.1 MeV in the welds and shells of the reactor pressure vessel that lie in the RPV beltline region. Neutron fluence is calculated at the end of operating cycle 17 and projected to the end of the reactor's design lifetime of 54 effective full power years (EFPY).

This fluence evaluation was performed in accordance with guidelines presented in U. S. Nuclear Regulatory Guide 1.190 [1]. In compliance with the guidelines, comparisons to flux wire measurements were performed to determine the accuracy of the RPV fluence model and an uncertainty analysis was performed to determine if a statistical bias exists in the model. Two separate flux wire activation analyses were conducted: one included flux wires irradiated from the start of plant operation through the end of cycle 6 and one included flux wires irradiated from the start of plant operation through the end of cycle 12. It was determined that the JAF fluence model does not have a statistical bias and that the results presented in this report are suitable for use in evaluating the effects of embrittlement on RPV material as specified in 10CFR50 Appendix G. The fluence presented in this report is for the reactor beltline region, which is defined in Appendices G and H of 10CFR50 as the areas of the RPV that exceed a fast neutron fluence (E>1.0 MeV) of 1.00E+17 n/cm^2.

The fluence values presented in this report were calculated using the RAMA Fluence Methodology [2]. The RAMA Fluence Methodology (hereinafter referred to as the Methodology) has been developed for the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in Boiling Water Reactor (BWR) components. The Methodology has been approved by the U. S. Nuclear Regulatory Commission [3] for application in accordance with U. S. Regulatory Guide 1.190 [1]. Benchmark testing has been performed using the Methodology for several surveillance capsule and reactor pressure vessel fluence evaluations. Results of these benchmark efforts show that the Methodology accurately predicts fluence in the RPV and surveillance capsule components of BWRs.

The information and associated evaluations provided in this report have been performed in accordance with the requirements of 10CFR50 Appendix B.

2
SUMMARY OF RESULTS

This section provides a summary of the results of the RPV fluence evaluation for JAF. The primary purpose of this evaluation is to determine the RPV fluence for energy >1.0 MeV and for energy >0.1 MeV at selected welds and shells in the RPV beltline region. Only the >1.0 MeV results are summarized here. Detailed results for >1.0 MeV are presented in Section 7. The >0.1 MeV results are discussed in Section 8 of this report. Peak neutron fluence is predicted for two points in time: at the end of cycle 17 (EOC 17) and projected to the end of the reactor's design life at 54 EFPY. Fluence is calculated at the inner surface (0T) and at 1/4T for each RPV weld and shell in the RPV beltline region.

Table 2-1 summarizes the peak fluence results for this evaluation for energy >1.0 MeV at EOC 17 (22.2 EFPY) and at 54 EFPY for the inner surface of the RPV welds and shells. One value reports the peak fluence for the weld locations and the other reports the peak fluence at the shell locations. The peak fluence for the weld locations is in circumferential weld VC-3-4 with a value of 2.53E+18 n/cm^2 at 54 EFPY. The peak fluence for the RPV shells is in the lower intermediate shell with a value of 3.11E+18 n/cm^2 at 54 EFPY.

Table 2-1
Peak Neutron Fluence for Energy >1.0 MeV for JAF RPV Weld and Shell Locations at the Inner Surface

Weld/Shell Location	Elevation [cm (in)]	Peak Fluence for EOC 17 (22.2 EFPY) (n/cm^2)	Peak Fluence for 54 EFPY (n/cm^2)
Weld VC-3-4[1] (43° azimuth)	642.62 (253.00)	1.31E+18	2.53E+18
Lower Intermediate Shell (43° azimuth)	720.25 (283.56)[2]	1.52E+18	----
Lower Intermediate Shell (43° azimuth)	781.21 (307.56)[2]	----	3.11E+18

1) Weld fluences are calculated at the center of the weld.
2) The peak fluence value occurred at a different elevation for EOC 17 than for 54 EFPY.

It was observed that the >1.0 MeV threshold fluence value of 1.00E+17 n/cm^2 was reached prior to the end of operating cycle 17 (22.2 EFPY) in a majority of the lower intermediate shell locations. Several locations in the lower shell exceed the threshold value, but no welds in the upper intermediate shell are expected to exceed 1.00E+17 n/cm^2 during the reactor's design life. The elevation ranges at which the fluence value exceeds 1.00E+17 n/cm^2 are given in Table 2-2. Based upon the elevation ranges shown in Table 2-2, the recirculation inlet or suction nozzles, including vessel to nozzle welds, are outside the RPV beltline region, and therefore, do not exceed fluence of 1.00E+17 n/cm^2 at 54 EFPY.

Table 2-2
RPV Fluence Threshold Elevation Range for JAF

Reactor Lifetime	Lower Elevation[1] [cm (inches)]	Upper Elevation[1] [cm (inches)]
EOC 17 (22.2 EFPY)	504.21 (198.51)	930.05 (366.16)
54 EFPY	493.57 (194.32)	948.50 (373.43)

1) Elevations are relative to the RPV zero elevation.

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Detailed results of the surveillance capsule activation analyses are presented in Section 5 of this report.

[[ ]]

In conclusion, it is determined that the RAMA Fluence Methodology produces accurate results that compare well with measured data. The Methodology for determining the neutron fluence for the RPV shell and weld locations has been performed in accordance with the guidelines presented in Regulatory Guide 1.190.

3
DESCRIPTION OF THE REACTOR SYSTEM

This section describes the JAF fluence model used in the RPV fluence evaluation. The fluence model is based on plant-specific design inputs including component mechanical designs, material compositions, and reactor operating history. Plant-specific mechanical design drawings and structural material data were provided by Entergy Nuclear Operations, Inc. and were used to build the JAF Nuclear Power Station RAMA fluence model [4]. Core simulator data was provided for cycles 1 through 18 [5].

3.1 Reactor System Mechanical Design Inputs

The JAF reactor is modeled with the RAMA Fluence Methodology. The Methodology employs a three-dimensional modeling technique to describe the reactor geometry for the neutron transport calculations. Detailed mechanical design information is used in order to build an accurate three-dimensional RAMA computer model of the reactor system.

The James A. FitzPatrick Nuclear Power Station is a General Electric BWR/4 class reactor with a core loading of 560 fuel assemblies. The rated thermal power output of the reactor was 2436 MWt for cycles 1-12. A power uprate was achieved in cycle 13 raising the thermal power to 2536 MWt.

Figure 3-1 illustrates the basic planar geometry configuration of the reactor at the axial elevation corresponding to the core mid-plane. All radial regions comprising the fluence model are illustrated. Beginning at the center of the reactor and projecting outwards, the regions include: the core region, including control rod locations and fuel assembly locations (fuel locations are shown only for the northeast quadrant); core reflector region (bypass water); central shroud wall; downcomer water region including the jet pumps; reactor pressure vessel (RPV) wall; mirror insulation; biological shield (concrete wall); and cavity regions between the RPV and biological shield. Also shown are the azimuthal positions of the surveillance capsules in the downcomer region at 30, 120, and 300 degrees and the jet pump assemblies at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees. Shroud repair tie rods were installed in the reactor during reload 11 and are azimuthally positioned at 15, 45, 75, 135, 165, 195, 225, 255, 315 and 345 degrees.

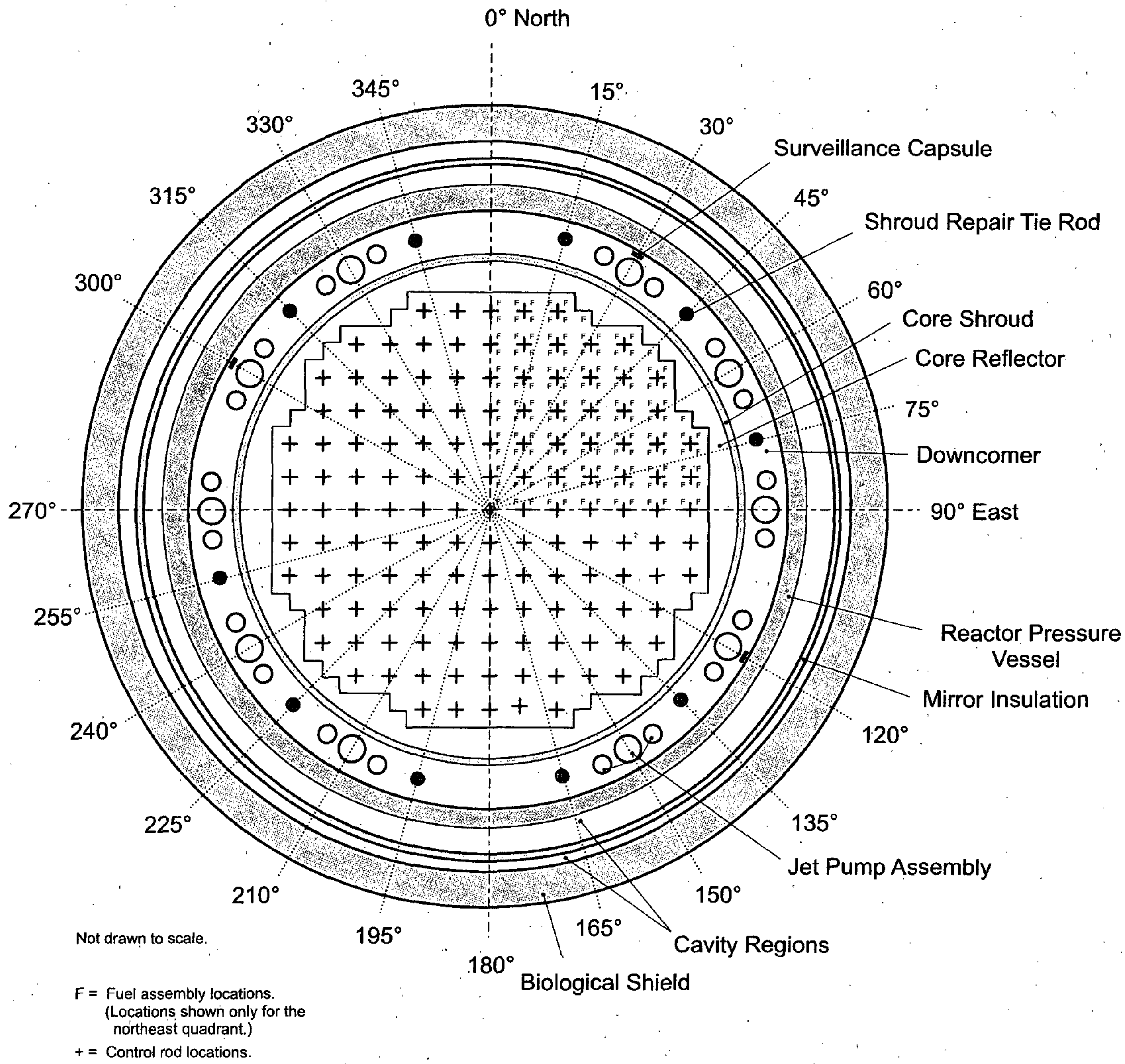

Figure 3-1
Planar View of JAF at the Core Mid-plane Elevation

The primary interest in this fluence evaluation is the determination of the neutron fluence at specified RPV welds and shells. Figure 3-2 identifies these specific weld locations. The vertical welds all begin with "VV" while the circumferential welds begin with "VC". The RPV weld fluence results are presented in this report using these weld identifications. [[

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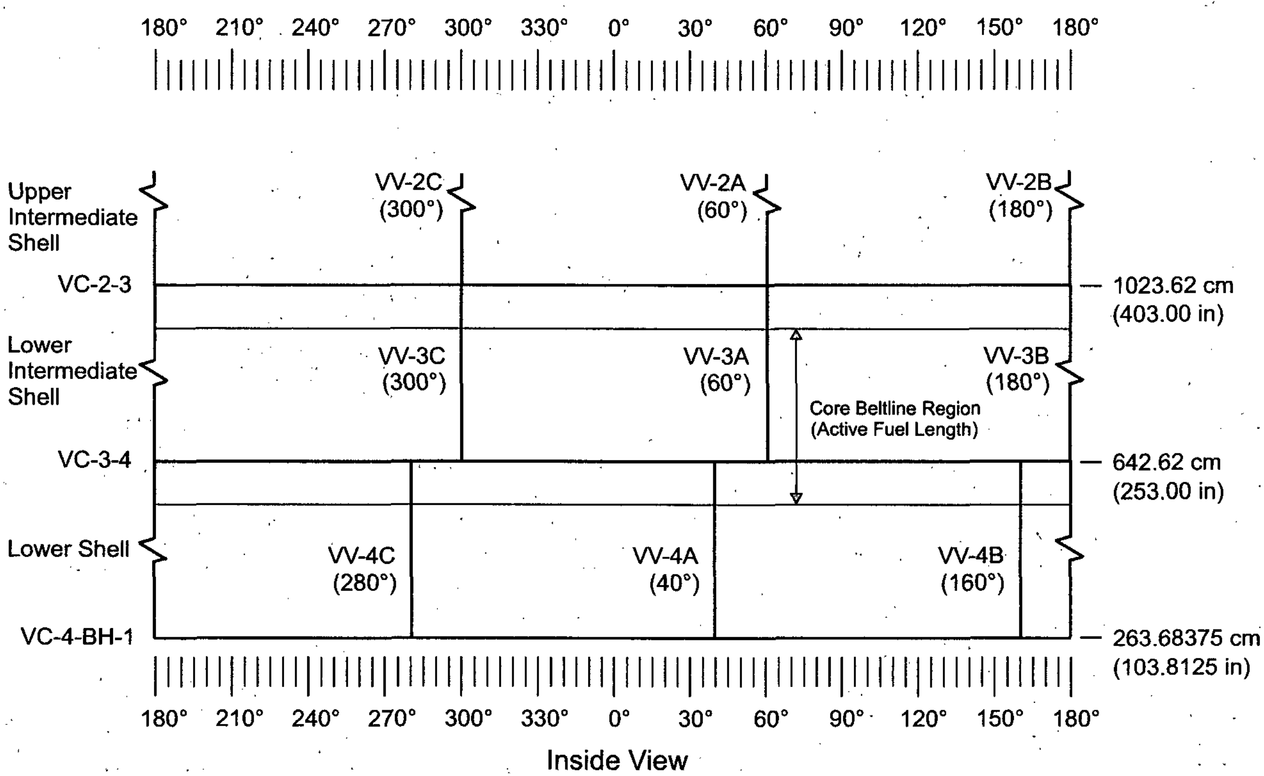

Figure 3-2
JAF RPV Shell Plates and Weld Location Identifiers

3.2 Reactor System Material Compositions

Each region of the reactor is comprised of materials that include reactor fuel, steel, water, insulation, concrete, and air. Accurate material information is essential for the fluence evaluation as the material compositions determine the scattering and absorption of neutrons throughout the reactor system and, thus, affect the determination of neutron fluence in the reactor components.

Table 3-1 provides a summary of the material compositions in the various components and regions of the JAF reactor. The attributes for the steel, insulation, and air compositions (i.e., material densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. The coolant water densities in the ex-core regions can vary with reactor heat balance through and between operating cycles[[ ]]. The attributes of the fuel compositions in the reactor core region change continuously during an operating cycle due to changes in power level, fuel burnup, control rod movements, and changing moderator density levels (voids). Because of the dynamics of the fuel attributes with reactor operation, one to several data sets are used to describe the operating states of the reactor core throughout each operating cycle. The number of data sets used in this analysis is presented in Section 3.3.2.

Table 3-1
Summary of Material Compositions by Region for JAF Nuclear Plant

Region	Material Composition
Control Rods and Guide Tubes	Stainless Steel and B_4C
Core Support Plate	Stainless Steel
Fuel Support Piece	Stainless Steel and B_4C
Fuel Bundle Lower Tie Plate	Stainless Steel, Zircaloy, Inconel
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zircaloy, Water
Core Reflector	Water
Fuel Bundle Upper Tie Plate	Stainless Steel, Zircaloy, Inconel
Top Guide	Stainless Steel
Core Spray Sparger Pipes	Stainless Steel
Core Spray Sparger Flow Areas	Water
Shroud	Stainless Steel
Shroud Repair Tie Rods[1]	Stainless Steel
Downcomer Region	Water
Jet Pump Riser and Mixer Flow Areas	Water
Jet Pump Riser and Mixer Metal	Stainless Steel
Surveillance Capsule Specimens	Carbon Steel
Reactor Pressure Vessel Clad	Stainless Steel
Reactor Pressure Vessel Wall	Carbon Steel
Cavity Regions	Air
Insulation	Aluminum, Stainless Steal
Biological Shield Clad	Carbon Steel
Biological Shield Wall	Reinforced Concrete

1) The shroud repair tie rods were introduced into the reactor during reload 11 (December 1994).

3.3 Reactor Operating Data Inputs

An accurate evaluation of fluence in the reactor requires an accurate accounting of the reactor operating history. The primary reactor operating parameters that affect neutron fluence evaluations for BWR's include the reactor power level, core power distribution, core void fraction distribution (or equivalently, water density distribution), and fuel material distribution.

3.3.1 Power History Data

The reactor power history used in the JAF RPV fluence evaluation was based on daily power history information for operating cycles 1 through 17 [5] and a projected equilibrium cycle for cycle 18 and beyond. [[

]] The power history data accounts for the reactor shutdown periods. The shutdowns were primarily due to the refueling outages between cycles. Table 3-2 provides the accumulated effective full power years of power generation at the end of each cycle in this fluence evaluation.

The rated thermal power output of JAF for operating cycles 1 through 12 is specified as 2436 MWt. A power uprate was achieved in cycle 13 raising the rated thermal power output to 2536 MWt. The power level 2536 MWt is used for projection purposes to the end of JAF's operating life.

3.3.2 Reactor State-Point Data

Reactor operating data for the JAF RPV fluence evaluation was provided as state-point data files for operating cycles 1 through 17 [5]. The state-point files provide a best-available representation of the operating conditions of the unit over the operating lifetime of the reactor. The data files include three-dimensional data arrays that describe the fuel materials, moderator materials, and the relative power distribution in the core region.

A separate neutron transport calculation was performed for each of the available state points. The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.3.1 in order to predict the neutron fluence in the reactor pressure vessel.

3.3.2.1 Beginning of Operation through Cycle 18 State Points

A total of [[ ]] state-point data files were used to represent the 17 reported operating cycles of JAF. [[

]] Table 3-2 shows the number of state points used for each cycle in this fluence evaluation.

Table 3-2
Number of State-point Data Files for Each Cycle in JAF

Cycle Number	Number of State Point Data Files	Rated Thermal Power MWt[2]	Accumulated Effective Full Power Years (EFPY)
1	[[	2436	1.3
2		2436	2.0
3		2436	2.8
4		2436	3.8
5		2436	4.9
6		2436	6.0
7		2436	7.4
8		2436	8.5
9		2436	9.7
10		2436	10.6
11		2436	12.0
12		2436	13.4
13		2536	15.1
14		2536	16.7
15		2536	18.6
16		2536	20.4
17		2536	22.2
18	]]	2536	24.0
>18	See note 1		54.0

1) Complete operating data for cycles 18 and beyond was not available at the time of the analysis. An equilibrium cycle (denoted as cycle 18) was developed by Entergy to represent the best available operating data for projecting fluence to the plant license end-of-life.
2) Although the rated thermal power level is listed for each cycle, actual power levels were used in modeling the plant's operation during each cycle.

3.3.2.2 Projected Operation through End of Design Life State Points

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reactor is licensed at a thermal power level of 2536 MWt for cycles beyond operating cycle 13. Therefore, fluence projections based on EFPY values use 2536 MWt as the basis for determining the amount of energy produced during one EFPY.

The projection of fluence to the end of the reactor licensed lifetime employs certain assumptions that can change. For example, if future reactor cycles deviate from the equilibrium cycle assumed in this analysis, then evaluations using the projected fluence may be inaccurate. Deviations from equilibrium cycle conditions can be incurred as the result of, for example, power uprates, new fuel designs, and revised heat balances. It is recommended that each future operating cycle be evaluated for potential impact on the projected fluence presented in this report and that the fluence analysis be updated accordingly.

3.3.3 Core Loading Pattern

It is common in BWRs that more than one fuel assembly design will be loaded in the reactor core in any given operating cycle. For fluence evaluations, it is important to account for the fuel assembly designs that are loaded in the core in order to accurately represent the neutron source distribution at the core boundaries (i.e., peripheral fuel locations, top fuel nodes, and bottom fuel nodes).

Six different basic fuel assembly designs are used in the reactor during cycles 1 through 18. Table 3-3 provides a summary of the fuel designs loaded in the reactor core for these operating cycles. The cycle core loading patterns are used to identify the fuel assembly designs in each cycle and their location in the core loading pattern. For each cycle, appropriate fuel assembly models are used to build the reactor core region of the JAF RAMA fluence model.

Table 3-3
Summary of the JAF Core Loading Pattern

Cycle	General Electric (GE) 7x7 Fuel Assembly Designs	General Electric (GE) 8x8 Fuel Assembly Designs	Westinghouse 8x8 Fuel Assembly Designs	General Electric (GE) 9x9 Fuel Assembly Designs	Framatome ANP 10x10 Fuel Assembly Designs	General Electric (GE) 10x10 Fuel Assembly Designs	Dominant Peripheral Fuel Design in the RAMA Model
1	560						GE 7x7
2	428	132					GE 7x7
3	292	268					GE 7x7
4	132	428					GE 7x7
5		560					GE 8x8
6		560					GE 8x8
7		560					GE 8x8
8		556	4				GE 8x8
9		556	4				GE 8x8
10		552	4	4			GE 8x8
11		404		156			GE 8x8
12		200		356	4		GE 8x8
13		12		352	4	192	GE 9x9
14				174	4	382	GE 9x9
15						560	GE 10x10
16						560	GE 10x10
17						560	GE 10x10
18						560	GE 10x10

4 CALCULATION METHODOLOGY

The JAF RPV fluence evaluation was performed using the RAMA Fluence Methodology software package [2] in compliance with the NRC Safety Evaluation Report [3]. The Methodology and the application of the Methodology to the JAF reactor are described in this section.

4.1 Description of the RAMA Fluence Methodology

The RAMA Fluence Methodology is a system of codes that is used to perform fluence evaluations in light water reactor components. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library [6]. The Methodology and procedures for its use are described in the following reports: Theory Manual [7] and Procedures Manual [8].

The primary inputs for the RAMA Fluence Methodology are mechanical design parameters and reactor operating history data. The mechanical design inputs are obtained from reactor design drawings (or vendor drawings) of the plant. The reactor operating history data is obtained from reactor core simulation calculations, system heat balance calculations, and daily operating logs that describe the operating conditions of the reactor.

The primary outputs from the RAMA Fluence Methodology calculations are neutron flux, neutron fluence, flux wire activations, and uncertainty determinations. The RAMA transport code calculates the neutron flux distributions that are used in the determination of neutron fluence. Several transport calculations are typically performed over the operating life of the reactor in order to calculate neutron flux distributions that accurately characterize the operating history of the reactor. The post-processing code (RAFTER) is then used to calculate component fluence and nuclide activations using the neutron flux solutions from the transport calculations and daily operating history data for the plant.

4.2 The RAMA Geometry Model for the JAF Nuclear Power Station

The RAMA Fluence Methodology uses a flexible three-dimensional modeling technique to describe the reactor geometry. The geometry modeling technique is based on the Cartesian coordinate system in which the (x,y) coordinates describe an axial plane of the reactor system and the z-axis describes elevations of the reactor system.

Figure 4-1 illustrates the planar configuration of the JAF model at an axial elevation near the core mid-plane of the reactor pressure vessel. In the radial dimension the model extends from the center of the RPV to the outside surface of the biological shield (401.0025 cm). Nine radial regions are defined in the JAF model: the core region (comprised of interior and peripheral fuel assemblies), core reflector, shroud, downcomer region, pressure vessel, mirror insulation, biological shield, and inner and outer cavity regions. The pressure vessel has cladding on the wall inner surface. The biological shield has cladding on the inner and outer surfaces. The downcomer region includes representations for the jet pumps, surveillance capsules, and, from cycle 12 onward, shroud repair tie rods.

Figure 4-1 shows that the reactor core region is modeled with rectangular geometry to preserve the shape of the core region. [[

]] The peripheral fuel assemblies are the primary contributors to the neutron source in the fluence calculation and are modeled to preserve the pin-wise source contribution at the core-core reflector interface.

Each of the components and regions that extend outward from the core region are modeled in their correct geometrical form. The core shroud, downcomer, RPV wall, mirror insulation, biological shield wall, and cavity regions are modeled as cylindrical parts. The shapes of other significant reactor components are appropriately represented in the model.

The surveillance capsule, which is rectangular in design, is modeled [[

]] and is correctly positioned behind the jet pump riser pipe at a radial position near the inner surface of the RPV wall. [[

]] Downcomer water surrounds the capsule on all sides.

The jet pump assembly design is modeled using cylindrical pipe elements for the jet pump riser and mixer pipes. The riser pipe is appropriately situated on a curvilinear path between the centers of the mixer pipes.

The shroud tie rod repair assembly is modeled [[

]] The tie rods are surrounded by downcomer water on all sides.

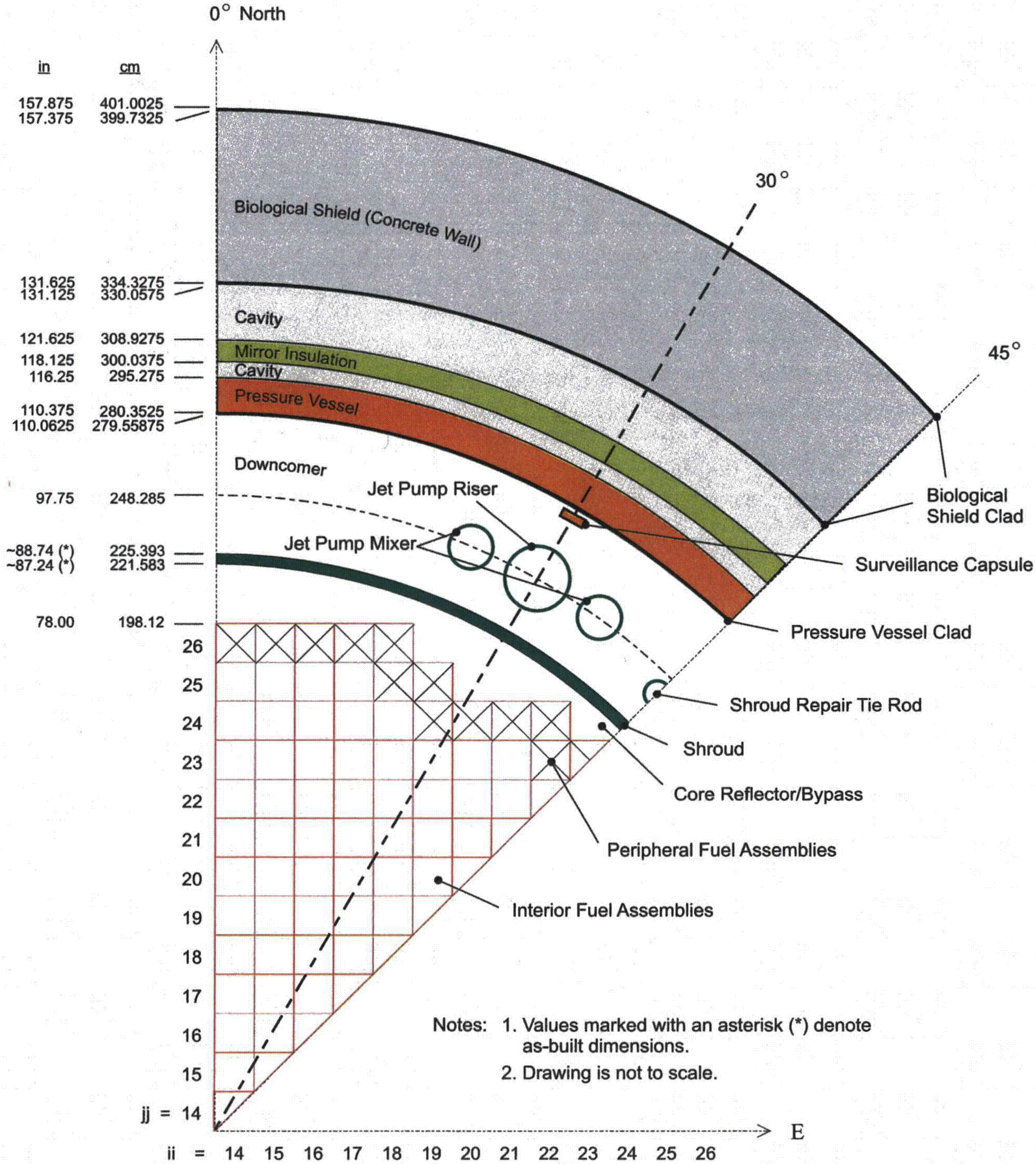

Figure 4-1
Planar View of the JAF RAMA Octant Model at the Core Mid-plane Elevation

For computational reasons, the RAMA model of the JAF reactor assumes octant symmetry. In the azimuthal dimension the model spans from 0 to 45 degrees where the 0 degree azimuth corresponds to the north compass direction that is specified in the reactor design drawings.

The jet pumps are shown as modeled at azimuth 30 degrees in the downcomer region. When symmetry is applied to the model, the eight jet pump assemblies that are positioned azimuthally at 30, 60, 120, 150, 210, 240, 300, and 330 degrees are represented by the model. Due to model symmetry restrictions, the jet pumps at the 90 and 270 degree azimuths are not represented. This is a conservative approximation for RPV fluence calculations.

The surveillance capsules are shown as modeled at azimuth 30 degrees. When symmetry is applied to the model, this location represents each of the surveillance capsules loaded at 30, 120, and 300 degrees (see Figure 3-1).

Figure 4-1 also shows that the shroud repair tie rods are represented at azimuth 45 degrees. When symmetry is applied to the model, this location represents each of the tie rods at 15, 45, 75, 135, 165, 195, 225, 255, 315 and 345 degrees in the reactor (see Figure 3-1).

Figure 4-2 provides an illustration of the axial configuration of the JAF RAMA model for four significant components: a fuel column, the core shroud, the downcomer region, and the reactor pressure vessel. Also shown in the figure is the relative axial positioning of the jet pumps, surveillance capsules, top guide, and core spray sparger pipes in the reactor model. The axial planes are divided into several groups representing particular component regions of the model as follows: the core region, the top guide, the shroud head flange, the core spray spargers, the fuel support piece, core support plate, and core inlet region. Sub-planar meshing is used in the model, as needed, to properly represent the positioning of reactor components, such as the surveillance capsules and jet pump rams head. [[

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The RPV horizontal (circumferential) welds analyzed in this evaluation are shown in Figure 4-2. Although the drawing is not precisely to scale, it provides a perspective of their locations in the reactor model relative to the core beltline region and other reactor components. Fluence is calculated at the elevations for the RPV circumferential welds depicted in the figure.

There are several key features of the RAMA code system that allow the reactor design to be accurately represented for reactor pressure vessel fluence evaluations. Following is a list of some of the key features of the model.

- Rectangular and cylindrical bodies are mixed in the model in order to provide an accurate geometrical representation of the components and regions in the reactor.

- The core geometry is modeled using rectangular bodies to represent the fuel assemblies in the reactor core region.

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Figure 4-2
Axial View of the JAF RAMA Model

- Cylindrical bodies are used to represent the components and regions that extend outward from the core region.

- A combination of rectangular and cylindrical bodies is used to describe the transition parts that are required to interface the rectangular core region to the cylindrical outer core regions.

- The top guide is appropriately modeled by including a representation of the vertical fuel assembly parts and top guide plates. The upper fuel assembly parts that extend into the top guide region are modeled in three axial segments: the fuel rod plenum, fuel rod upper end plugs, and fuel assembly upper tie plate.

- The jet pump assembly model includes representations of the riser, mixer, and diffuser pipes; nozzles; and riser brace yoke, leafs, and pads.

- The fuel support piece, core support plate, and core inlet regions appropriately include a representation of the cruciform control rod below the core region. The lower fuel assembly parts include representations for the fuel rod lower end plugs, lower tie plate, and nose piece.

- The core support plate includes detailed representation of the core plate plugs.

- The surveillance capsules are represented in the downcomer region at the correct azimuth, at an axial elevation corresponding to the core mid-plane elevation, and radially near the inner surface of the pressure vessel wall.

- The G03 shroud repair tie rods are appropriately represented at the correct azimuth, radial, and axial locations in the downcomer region.

- The core spray spargers are appropriately represented as toruses in the model. The sparger pipes and nozzles reside inside the upper shroud wall above the top guide. The sparger model includes a representation of reactor coolant inside the pipes.

4.3 RAMA Calculation Parameters

The RAMA transport code uses a three-dimensional deterministic transport method to calculate neutron flux distributions in reactor problems. [[

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The RAMA transport calculation also uses information from the RAMA nuclear data library to determine the scope of the flux calculation. This information includes the Legendre expansion of the scattering cross sections that is used in the treatment of anisotropy of the problem. By default, the RAMA transport calculation uses the maximum order of expansion that is available for each nuclide in the RAMA nuclear data library (i.e., through P_5 scattering for actinide and zirconium nuclides and through P_7 scattering for all other nuclides in the model).

The neutron flux is calculated using an iterative technique to obtain a converged solution for the problem. [[
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4.4 RAMA Neutron Source Calculation

The neutron source for the RAMA transport calculation is calculated using the input relative power density factors for the different fuel regions and data from the RAMA nuclear data library.

The core neutron source is determined using the cycle-specific three-dimensional burnup distributions. The radial power gradient in the peripheral fuel assemblies is modeled to account for the pin-wise source distributions in the peripheral and inside corner fuel assemblies.

4.5 RAMA Fission Spectra

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5 SURVEILLANCE CAPSULE ACTIVATION AND FLUENCE RESULTS

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]] This section addresses the evaluation of the JAF surveillance capsule flux wires and the comparison to measurements. Two surveillance capsule activation analyses were performed for the JAF reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 6 and at the end of cycle 12. The cycle 6 flux wires were irradiated for 5.986 EFPY. The cycle 12 flux wires were irradiated for 13.4 EFPY. Details of the dosimetry specimens and analyses are presented in Section 5.1.

5.1 Comparison of Predicted Activation to Plant-Specific Measurements

5.1.1 Cycle 6 Capsule Activation Comparison Results

Three copper, three iron, and three nickel flux wires were irradiated in the JAF surveillance capsule during the first six cycles of operation. The flux wires were loaded in the flux wire holder attached to the 30 degree surveillance capsule container. Activation measurements were performed following irradiation for the following reactions [9]: $^{63}Cu(n,\alpha)^{60}Co$, $^{54}Fe(n,p)^{54}Mn$, and $^{58}Ni(n,p)^{58}Co$. [[

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5.1.2 Cycle 12 Capsule Activation Comparison Results

Two copper, two iron, and two nickel flux wires were irradiated in the JAF surveillance capsule during the first 12 cycles of operation. The flux wires were loaded in the flux wire holder attached to the 120 degree surveillance capsule container. Activation measurements were performed following irradiation for the following reactions [10]: $^{63}Cu(n,\alpha)^{60}Co$, $^{54}Fe(n,p)^{54}Mn$, and $^{58}Ni(n,p)^{58}Co$. Note that only the average measurements for copper, iron and nickel were provided in [10]. [[

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6
REACTOR PRESSURE VESSEL UNCERTAINTY ANALYSIS

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7
CALCULATED REACTOR PRESSURE VESSEL NEUTRON FLUENCE FOR ENERGY >1.0 MeV

The neutron fluence for the JAF RPV at the inner surface (0T) and at 1/4T is determined by the RAMA Fluence Methodology for two points in time: at the end of cycle 17 (22.2 EFPY) and projected to the design life of 54 EFPY. The results of the fluence evaluation for energy >1.0 MeV are presented in the tables that follow. As reported in the uncertainty analysis (see Section 6), the calculated pressure vessel fluence requires no adjustment to account for bias effects. The location and identification of the RPV welds are shown in Figure 3-2.

Tables 7-1 through 7-3 report the maximum >1.0 MeV fluence at 22.2 EFPY and 54 EFPY in the RPV weld locations for the RPV beltline region. The RPV beltline region is divided into three shells for the purposes of reporting these fluence results. Table 7-1 reports the fluence for the upper intermediate shell welds. Table 7-2 reports the fluence for the lower intermediate shell welds. Table 7-3 reports the fluence for the lower shell welds. The maximum fluence for the RPV welds is at the inner surface of circumferential weld VC-3-4 with a value of 2.53E+18 n/cm^2 at 54 EFPY.

Tables 7-4 through 7-6 report the maximum >1.0 MeV fluence in the RPV shells in the RPV beltline region at 22.2 EFPY and 54 EFPY. Tables 7-4 through 7-6 represent the same shell regions as used in Tables 7-1 through 7-3, respectively. The maximum fluence for the RPV shells is at the inner surface of the lower intermediate shell with a value of 3.11E+18 n/cm^2 at 54 EFPY. It is observed in Tables 7-4 through 7-6 that the peak fluence shifts azimuthally and axially in the RPV wall with time. This is attributed primarily to changes in reactor power distributions that are caused by power uprates, new fuel designs, and new reactor operating strategies.

It is observed that the threshold fluence value of 1.00E+17 n/cm^2 was reached prior to the end of operating cycle 17 (22.2 EFPY) at the inner surface in a majority of the lower intermediate shell and lower shell locations. No welds in the upper intermediate shell are expected to exceed 1.00E+17 n/cm^2 during the reactor's design life. The elevation range at which the fluence value exceeds 1.00E+17 n/cm^2 is determined to be 504.21 cm (198.51 inches) to 930.05 cm (366.16 inches) for 22.2 EFPY and 493.57 cm (194.32 inches) to 948.50 cm (373.43 inches) for 54 EFPY.

Table 7-1
Maximum >1.0 MeV Neutron Fluence in JAF Upper Intermediate RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 1/4T R= 284.08 cm (111.84 in)	54 EFPY Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	54 EFPY Fluence (n/cm^2) 1/4T R= 284.08 cm (111.84 in)
Upper Intermediate	VV-2A	3.96E+15	2.85E+15	1.04E+16	7.45E+15
	VV-2B	3.65E+15	2.70E+15	9.66E+15	7.08E+15
	VV-2C	3.96E+15	2.85E+15	1.04E+16	7.45E+15
	VC-2-3	4.11E+15	2.98E+15	1.07E+16	7.72E+15

Table 7-2
Maximum >1.0 MeV Neutron Fluence in JAF Lower Intermediate RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 1/4T R= 284.08 cm (111.84 in)	54 EFPY Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	54 EFPY Fluence (n/cm^2) 1/4T R= 284.08 cm (111.84 in)
Lower Intermediate	VV-3A	8.36E+17	5.75E+17	1.75E+18	1.20E+18
	VV-3B	7.08E+17	4.91E+17	1.49E+18	1.03E+18
	VV-3C	8.36E+17	5.75E+17	1.75E+18	1.20E+18
	VC-2-3	4.11E+15	2.98E+15	1.07E+16	7.72E+15
	VC-3-4	1.31E+18	8.87E+17	2.53E+18	1.71E+18

Table 7-3
Maximum >1.0 MeV Neutron Fluence in JAF Lower RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
		0T R= 280.75 cm (110.38 in)	1/4T R= 284.72 cm (112.09 in)	0T R= 280.75 cm (110.38 in)	1/4T R= 284.72 cm (112.09 in)
Lower	VV-4A	1.19E+18	7.55E+17	2.34E+18	1.48E+18
	VV-4B	7.89E+17	5.01E+17	1.63E+18	1.03E+18
	VV-4C	7.33E+17	4.67E+17	1.48E+18	9.45E+17

Table 7-4
Maximum >1.0 MeV Neutron Fluence in JAF Upper Intermediate RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
Shell	Azimuth (degrees)	Elevation [cm (in)]	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)
Upper Intermediate	37[1]	1023.62 (403.00)	4.11E+15			
	34[1]	1023.62 (403.00)		2.98E+15		7.72E+15
	36[1]	1023.62 (403.00)			1.07E+16	

1) The peak fluence value occurred at a different azimuth for the 0T location than for the 1/4T location at both EOC 17 and 54 EFPY.

Table 7-5
Maximum >1.0 MeV Neutron Fluence in JAF Lower Intermediate RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm²)		54 EFPY Fluence (n/cm²)	
Shell	**Azimuth (degrees)**	**Elevation [cm (in)]**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.08 cm (111.84 in)**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.08 cm (111.84 in)**
Lower Intermediate	43[1]	720.25[2] (283.56)	1.52E+18			
	44[1]	731.28[2] (287.90)		1.03E+18		
	43[1]	781.21[2] (307.56)			3.11E+18	
	43[1]	781.94[2] (307.85)				2.10E+18

1) The peak fluence value occurred at a different azimuth for the 0T location than for the 1/4T location at EOC 17.
2) The peak fluence value occurred at a different elevation for the 0T location than for the 1/4T location at both EOC 17 and 54 EFPY.

Table 7-6
Maximum >1.0 MeV Neutron Fluence in JAF Lower RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm²)		54 EFPY Fluence (n/cm²)	
Shell	**Azimuth (degrees)**	**Elevation [cm (in)]**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.72 cm (112.09 in)**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.72 cm (112.09 in)**
Lower	43[1]	642.62 (253.00)	1.31E+18		2.53E+18	1.71E+18
	44[1]	642.62 (253.00)		8.87E+17		

1) The peak fluence value occurred at a different azimuth for the 0T location than for the 1/4T location at EOC 17.

8
CALCULATED REACTOR PRESSURE VESSEL NEUTRON FLUENCE FOR ENERGY >0.1 MeV

The neutron fluence for the JAF RPV at the inner surface (0T) and at 1/4T is determined by the RAMA Fluence Methodology for two points in time: at the end of cycle 17 (22.2 EFPY) and projected to the design life of 54 EFPY. The results of the fluence evaluation for energy >0.1 MeV are presented in the tables that follow.

Tables 8-1 through 8-3 report the maximum >0.1 MeV fluence at 22.2 EFPY and 54 EFPY in the RPV weld locations for the RPV beltline region. The RPV beltline region is divided into three shells for the purposes of reporting these fluence results. Table 8-1 reports the fluence for the upper intermediate shell welds. Table 8-2 reports the fluence for the lower intermediate shell welds. Table 8-3 reports the fluence for the lower shell welds. The maximum fluence for the RPV welds is at the inner surface of circumferential weld VC-3-4 with a value of 4.98E+18 n/cm^2 at 54 EFPY.

Tables 8-4 through 8-6 report the maximum >0.1 MeV fluence in the RPV shells in the RPV beltline region at 22.2 EFPY and 54 EFPY. Tables 8-4 through 8-6 represent the same shell regions as used in Tables 8-1 through 8-3, respectively. The maximum fluence for the RPV shells is at the inner surface of the lower intermediate shell with a value of 6.03E+18 n/cm^2 at 54 EFPY. It is observed in Tables 8-4 through 8-6 that the peak fluence shifts azimuthally and axially in the RPV wall with time. This is attributed primarily to changes in reactor power distributions that are caused by power uprates, new fuel designs, and new reactor operating strategies.

Table 8-1
Maximum >0.1 MeV Neutron Fluence in JAF Upper Intermediate RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
		0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)
Upper Intermediate	VV-2A	1.05E+16	1.16E+16	2.65E+16	2.86E+16
	VV-2B	9.83E+15	1.14E+16	2.49E+16	2.81E+16
	VV-2C	1.05E+16	1.16E+16	2.65E+16	2.86E+16
	VC-2-3	1.10E+16	1.21E+16	2.74E+16	2.94E+16

Table 8-2
Maximum >0.1 MeV Neutron Fluence in JAF Lower Intermediate RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
		0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)
Lower Intermediate	VV-3A	1.73E+18	1.50E+18	3.63E+18	3.14E+18
	VV-3B	1.39E+18	1.24E+18	2.91E+18	2.60E+18
	VV-3C	1.73E+18	1.50E+18	3.63E+18	3.14E+18
	VC-2-3	1.10E+16	1.21E+16	2.74E+16	2.94E+16
	VC-3-4	2.58E+18	2.25E+18	4.98E+18	4.34E+18

Table 8-3
Maximum >0.1 MeV Neutron Fluence in JAF Lower RPV Vertical and Circumferential Welds

Shell Location	Weld	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	EOC 17 (22.2 EFPY) Fluence (n/cm^2) 1/4T R= 284.72 cm (112.09 in)	54 EFPY Fluence (n/cm^2) 0T R= 280.75 cm (110.38 in)	54 EFPY Fluence (n/cm^2) 1/4T R= 284.72 cm (112.09 in)
Lower	VV-4A	2.37E+18	2.01E+18	4.64E+18	3.93E+18
	VV-4B	1.56E+18	1.33E+18	3.22E+18	2.75E+18
	VV-4C	1.44E+18	1.23E+18	2.91E+18	2.50E+18

Table 8-4
Maximum >0.1 MeV Neutron Fluence in JAF Upper Intermediate RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
Shell	Azimuth (degrees)	Elevation [cm (in)]	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)	0T R= 280.75 cm (110.38 in)	1/4T R= 284.08 cm (111.84 in)
Upper Intermediate	38[1]	1023.62 (403.00)	1.10E+16			
	37[1]	1023.62 (403.00)		1.21E+16	2.74E+16	
	34[1]	1023.62 (403.00)				2.94E+16

1) The peak fluence value occurred at a different azimuth for the 0T location than for the 1/4T location at both EOC 17 and 54 EFPY.

Table 8-5
Maximum >0.1 MeV Neutron Fluence in JAF Lower Intermediate RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
Shell	**Azimuth (degrees)**	**Elevation [cm (in)]**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.08 cm (111.84 in)**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.08 cm (111.84 in)**
Lower Intermediate	44[1]	730.41[2] (287.56)	2.96E+18			
	44[1]	728.74[2] (286.90)		2.56E+18		
	43[1]	781.21[2] (307.56)			6.03E+18	
	44[1]	781.94[2] (307.85)				5.20E+18

1) The peak fluence value occurred at a different elevation for the 0T location than for the 1/4T location at EOC 17.
2) The peak fluence value occurred at a different azimuth and different elevation for the 0T location than for the 1/4T location at both EOC 17 and 54 EFPY.

Table 8-6
Maximum >0.1 MeV Neutron Fluence in JAF Lower RPV Shell

Shell Plate Location			EOC 17 (22.2 EFPY) Fluence (n/cm^2)		54 EFPY Fluence (n/cm^2)	
Shell	**Azimuth (degrees)**	**Elevation [cm (in)]**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.72 cm (112.09 in)**	**0T R= 280.75 cm (110.38 in)**	**1/4T R= 284.72 cm (112.09 in)**
Lower	44[1]	642.62 (253.00)	2.58E+18	2.25E+18		4.34E+18
	43[1]	642.62 (253.00)			4.98E+18	

1) The peak fluence value occurred at a different azimuth for the 0T location than for the 1/4T location at 54 EFPY.

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REFERENCES

1. "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Nuclear Regulatory Commission Regulatory Guide 1.190, March 2001.

2. *BWRVIP-126: BWR Vessel Internals Project, RAMA Fluence Methodology Software, Version 1.0*, EPRI, Palo Alto, CA: 2003. 1007823.

3. "Letter from William H. Bateman (U. S. NRC) to Bill Eaton (BWRVIP), "Safety Evaluation of Proprietary EPRI Reports BWRVIP-114, -115, -117, and -121 and TWE-PSE-001-R-001," dated May 13, 2005.

4. "James A. FitzPatrick Geometry Inputs," TransWare Enterprises Inc., ENT-FLU-002-R-003, Rev. 0, September 2007.

5. "CASMO-4/SIMULATE-3 Fitzpatrick Cycles 1-18 Data Calculations for RAMA Fluence Methodology," Entergy Nuclear Operations, Inc., JAF-CALC-07-00031, Rev. 0, October 2007.

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10. T. J. Griesbach, *Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY*, General Electric Company, GE-NE-B1100732-01, Rev. 1, February 1998.

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Attachment 5

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 14

Axial Weld Analysis Clarification

Axial Weld Analysis Clarification

Section 4.2.6 Reactor Vessel Axial Weld Failure Probability

Table 4.2-5 compares the JAFNPP reactor vessel limiting axial weld parameters to those used in the NRC analysis. ~~The data in the second column (CEOG 32 EFPY) is from Table 2.6-4 of the NRC SER for BWRVIP-05 (Reference 4.2-7). The data in the third column is based on the projected 32 EFPY fluence for JAFNPP and the limiting weld chemistry.~~ The data in the middle~~fourth~~ column is from Table 2.6-5 of the NRC SER for BWRVIP-05 and Table 1 of the NRC SER for BWRVIP-74 (Reference 4.2-7, 4.7-1). The data in the ~~last~~ right column is the projected 54 EFPY data for JAFNPP taken from Table 4.2-3. (For consistency with the NRC data, the JAF 54 EFPY~~Columns 2 and 3~~, ~~the EOL~~ mean RT_{NDT} is calculated without margin and hence is lower than the Table 4.2-3 RT_{NDT} value.)

Table 4.2-5 is replaced with the following.

Table 4.2-5
Effects of Irradiation on JAFNPP RPV Axial Weld Properties

Plant / Parameter Description	NRC Limiting Plant-Specific Data	JAFNPP Data for Weld 2-233A/B/C
EFPY	NA	54
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-2[1]	-48
Neutron fluence, n/cm[2]	1.48E+18[3]	2.34E+18
Fluence factor (FF) (calculated per RG1.99 based on fluence in previous line)	0.500[4]	0.608
Weld copper content, %	0.219[3]	0.219
Weld nickel content, %	0.996[3]	0.996
Weld chemistry factor (CF)	231.1[4]	231.1
Fluence factor times chemistry factor (FF x CF)	116[2]	140.4
Margin (implied), °F	0.0	0.0
Increase in reference temperature (ΔRT_{NDT}), °F (FF x CF + Margin)	116[2]	140.4
Mean adjusted reference temperature (ART), °F (RT_{NDT} + ΔRT_{NDT}T)	114[1]	92.4

[1] Taken from Table 1 of the Safety Evaluation Report for BWRVIP-74. (Also in Table 3 of the supplemental SER for BWRVIP-05)
[2] The mean adjusted reference temperature minus the initial reference temperature
[3] Taken from the supplemental SER for BWRVIP-05
[4] Determined using tables and formula from Regulatory Guide 1.99

Axial Weld Analysis Clarification

Section 4.2.7 References

Add the following reference.

4.2-8 Strosnider, J. R. (NRC) to C. Terry (BWRVIP Chairman), "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project, BWRVIP-05 Report (TAC NO. MA3395)" letter dater March 7, 2000.

LRA Section A.2.2.1.6, Reactor Vessel Axial Weld Failure Probability

The second paragraph is replaced with the following.

The BWRVIP-74 SER states it is acceptable to show that the mean RT_{NDT} of the limiting beltline axial weld at the end of the period of extended operation is less than the limiting value (114 °F) given in Table 1 of the BWRVIP-74 SER. This value supports the axial weld failure probability and is based on the assumption of essentially 100% (> 90%) inspection of the axial welds in the beltline region. Due to various obstructions within the reactor vessel, JAFNPP is able to inspect approximately 88% of the axial welds in the beltline region. The NRC granted a relief request for less than 90% coverage. The projected 54 EFPY mean RT_{NDT} value for JAFNPP is well below the limiting mean RT_{NDT} of 114 °F. The 2% difference in the amount of inspected weld will not offset the 21.6 °F margin between the 92.4 °F mean RT_{NDT} for JAFNPP and the 114 °F mean RT_{NDT} used in the NRC SER for BWRVIP-74. Therefore, the axial weld failure probability will not exceed 5 E -6 per reactor operating year during the period of extended operation. As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Attachment 6

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 14

Supplemental Information for RAI 4.3.3-1

Supplemental Information for RAI 4.3.3-1

James A. FitzPatrick Nuclear Power Plant (JAFNPP) will comply with Commitment #20 as part of the Fatigue Monitoring Program (FMP) in accordance with 10 CFR 54.21(c)(1)(iii). This is accomplished as follows.

(1) NUREG-1801, Program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary, Element 4, Detection of Aging Effects, states "The program provides for periodic update of the fatigue usage calculations." Commitment #20 to refine the current fatigue analyses to include the effects of reactor water environment specifies details for updating the fatigue usage calculations consistent with this element (Element 4) of NUREG-1801, X.M1. The refined analyses will be accomplished as described in Commitment #20 with the clarifying details described in Attachment 1 of LRA Amendment 13, dated August 14, 2007 (reference JAFP 07-0100).

(2) NUREG-1801, Program X.M1, Element 7, Corrective Actions, states "The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation. Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation. For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations." Commitment #20 includes corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation consistent with this element (Element 7) of NUREG-1801, X.M1. Those corrective actions are the three options specified in the commitment for fatigue management if ongoing monitoring indicates a potential for a condition outside the analyses bounds.

The LRA and subsequent amendments treated the actions specified under Commitment #20 as separate from the JAFNPP Fatigue Monitoring Program and took exception to the consideration of reactor water environment in the program. Combining Commitment #20 with the FMP combines activities that the staff has reviewed as separate items. No new technical information or activities are introduced by combining these items. By considering the actions specified in Commitment #20 part of the FMP, the FMP becomes consistent with NUREG-1801 with no exceptions.